Design Issues on Compact Low Aspect Ratio DEMO Reactor, SlimCS

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Abstract. Recent design study on SlimCS focused mainly on the torus configuration including blanket, divertor, materials and maintenance scheme. For vertical stability of elongated plasma and high beta access, a sector-wide conducting shell is arranged in between replaceable and permanent blanket. The reactor adopts pressurized-water-cooled solid breeding blanket. Compared with an advanced concept with supercritical water in the previous DEMO (DEMO-2001), the design options satisfying tritium self-sufficiency are relatively scarce. Considered divertor technology and materials, an allowable heat load to the divertor plate should be 8 MW/m^2 or lower, which will be a critical constraint for determining a handling power of DEMO (namely, a summation of alpha heating power and external input power for current drive).

1. Introduction

In the primary conceptual study on DEMO, we proposed a compact low aspect ratio (low-A) DEMO reactor named “SlimCS” from a consideration of what DEMO concept can demonstrate economic viability of fusion [1]. The subsequent conceptual study in recent years reveals difficulties in ensuring consistency across different technologies relevant to DEMO. It is widely known that high power handling, tritium self-sufficiency, higher beta, toughness against severe irradiation and electromagnetic forces, and maintainability are themselves difficult issues. Obviously, design integration on DEMO, which is required to resolve all these issues systematically, is not simply a tradeoff problem but a formidable problem intricately intertwined with constraints of the relevant technologies. This paper reports DEMO design issues that we are facing in the design study of SlimCS.

SlimCS has a major radius of 5.5 m, aspect ratio of 2.6, maximum field of 16.4 T, normalized beta (β_N) of 4.3 and fusion output of 2.95 GW. The reactor is characterized by a reduced-size central solenoid (CS) whose main function is plasma shaping rather than plasma current ramp-up. The CS has an outer radius of 0.7 m, being capable of moderate plasma shaping (triangularity of ~0.35) and plasma current ramp of 3.8 MA. Although such a CS provides a constraint in tokamak operation, especially in the current ramp-up phase, it has advantages to allow us to introduce a thin toroidal coil system,

TABLE 1: Main design parameters of SlimCS

Major radius, R_p (m)	5.5	Normalized beta, β_N	4.3
Minor radius, a (m)	2.1	Stored energy, W_{tot} (MJ)	1,164
Aspect ratio, A	2.6	Temperature, $<T_e>$ (keV)	17.0
Plasma current, I_p (MA)	16.7	Density, $<n_e>$ (10^{20} m^{-3})	1.15
Toroidal field, B_T (T)	6.0	Normalized density, n_e/n_{GW}	1.0
Maximum field, B_{max} (T)	16.4	Confinement enhancement, HHy2	1.3
Elongation, κ_{95}	2.0	Bootstrap current fraction, f_{BS} (%)	~75
Triangularity, δ_{95}	0.35	Current drive power, P_{CD} (MW)	60-100
Safety factor, q_{95}	5.4	Fusion output, P_{fus} (MW)	2,950
Plasma volume, V_p (m^3)	941	Neutron wall load, P_n (MW/m^2)	~3

decreasing the reactor weight and perhaps contributing a reduction of the construction cost. In addition, the reduced-size CS produces the possibility of low A, which leads to advantages in physics design such as high elongation of plasma, high plasma current, high Greenwald density limit and high beta limit. TABLE 1 lists the main design parameters of the reactor.
In this paper, we divide the design issues on SlimCS into four; torus configuration, blanket, divertor and current drive. Approaches to resolve the design issues and unresolved problems are presented in the following sections.

2. Torus configuration

2.1 Conducting shell for high beta access

Taking the advantage of high β access in low A, SlimCS was designed to have β_N of 4.3 on the basis of a scaling law by Ref. [2]. In order to validate the design value, the β_N limit was recently calculated using the ERATO-J code. As shown in FIG.1, the β_N limit is 5.0 for bootstrap-dominated reversed shear plasma with q_{min} = 2.5-3 when an ideal conducting shell located at r_{shell}/a = 1.32. Based on this result, we determined to arrange the sector-wide conducting shell assembly at r_{shell}/a = 1.32 (FIG.2 (a)). For plasma with more gentle pressure gradient, higher β_N limit is expected and thus more adequate design margin will be provided. Another calculation for plasma with a moderate edge pedestal, which is favorable for steady state operation because of bootstrap current driven in the edge region where external current drive (CD) is less efficient due to low CD efficiency, provided the similar critical β_N. In order to clarify target plasma profiles of DEMO, this kind of beta limit study is necessary to be coupled with an assessment of steady state operation scenario in further study.

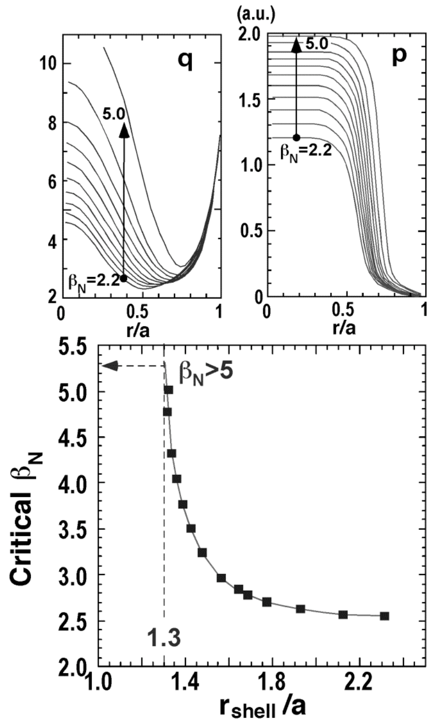


FIG.1. Calculated β_N limit for toroidal mode number of n = 1. Also shown are safety factor (q) and plasma pressure (p) profiles.

In SlimCS, 360° assembly of saddle shaped conducting shells is placed at r_{shell}/a = 1.30. In addition to high beta access, the shell assembly has a function of suppressing vertical instability of plasma. Although the requirement for the shell position makes the torus configuration somewhat complex, this problem is resolved by arranging the shell in between the 0.3-m thick replaceable blanket and 0.5-m thick permanent one. Originally, the shell was planned to use a 1-cm copper plate. However, because of difficulty of bonding the Cu plate to the poloidal ring structure (namely, permanent blanket and shield) made of reduced activation ferritic martensitic (RAFM) steel, the design was modified to have the poloidal ring structure provide

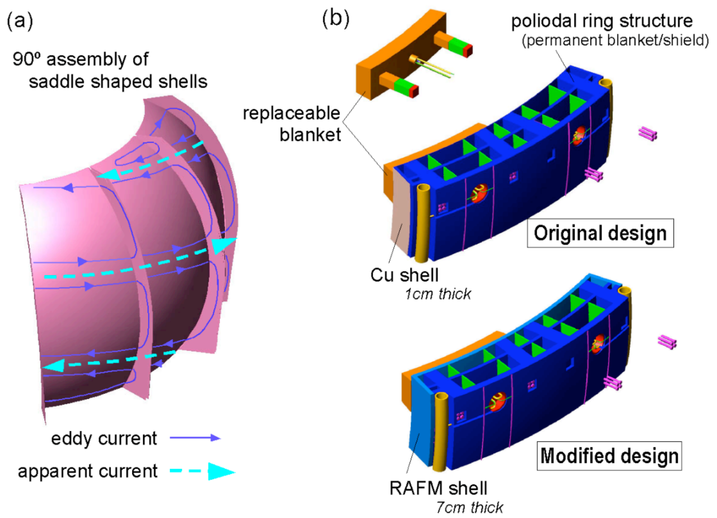


FIG.2. (a) Role of saddle shaped conducting shells and (b) arrangement of the shell in between the replaceable blanket and the poloidal ring structure (permanent blanket/shield).

such a shell function. For this purpose, the poloidal ring structure is composed of 7cm front and side plates as illustrated in FIG.2 (b). A concern of the modified design is a reduction of tritium breeding ratio (TBR) in the permanent blanket. However, a neutronics calculation indicates a required TBR can be obtained for $r_{shell}/a > 1.3$.
Considering the difficulty in the installation of resistive wall mode (RWM) stabilization coils in the vacuum vessel of the reactor where breeding blankets are set up in the most portion of the outboard, we regard RWM stabilization by plasma rotation as a realistic option.

2.2 Maintenance

Maintenance scheme is one of critical issues for DEMO. SlimCS adopts the sector transport hot cell maintenance scheme taking into account: 1) high availability, 2) flexibility for access to core components, and 3) extensibility for upgrading blanket. Figure 3 illustrates the concept of sector transport and the cask. A design philosophy is to conceive a maintenance scheme with the use of the existing or forseeeable technologies. In the sector maintenace scheme, the number of in-situ cutting/re-weldng points of piping is minimized. In addition, use of spare sectors minimizes the time required for the maintetance because the most time-consuming procceses such as re-welding and its inspection can be carried out in the hot cell during tokamak operation. The cask has double seal doors so that a cryostat port is sealed with one of the doors when the cask is undocked for the sector transport. There are two options for cask transfering mechanism. One is the carrier composed of wheels and roller bearings (FIG.3 (b)). A sector with a weight of 750 tons can be transported with the existing technologies. The cask runs on rails and change direction with turntables installed on the floor. Hovering is alternative transfering mechanism.

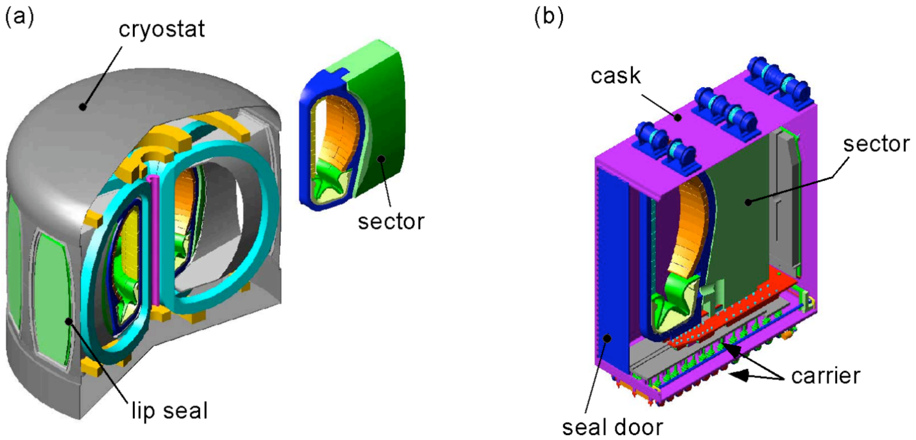


FIG.3. (a) Concept of sector maintenance and (b) cask delivering the sector.

Although the time required for replacing the whole replaceable blankets is dependent on peripheral equipments, our assessment indicates that the time for the sector transport scheme will be 2 months assuming that the most time-determining processes are the cutting/re-welding for piping, the lip seal of each sector and the removal/re-arrangement of the current drive (ECCD) system. In contrast, the time for in-vessel maintenance is estimated to be 8 months at least.
A critical design issue for sector maintenance is how to support an enormous turnover force of TF coils. In the case of in-vessel maintenance scheme like ITER, the turnover

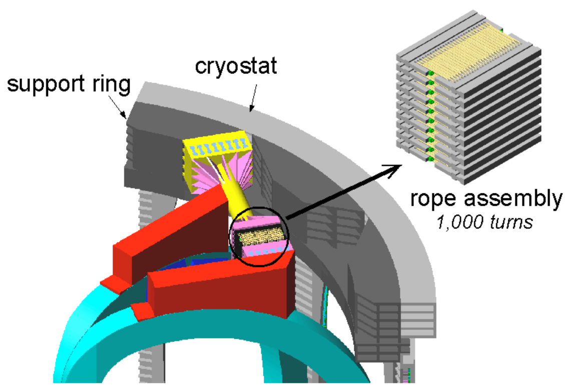


FIG.4. Concept of support for TF coil turnover force.

force is supported by inter-coil structure. In contrast, large open ports specific to the sector maintenance scheme do not allow us to setup inter-coil structure. In SlimCS, it is estimated that the turnover force is 10,000 tons/coil and the maximum deflection of TF coil without supports is 1.25 m in the toroidal direction. In order to support the turnover force, a support structure with the use of tension force of ropes (FIG.4) is proposed. Merits of the concept are ease of balanced loading and length tuning of support. Support of the resulting torsion of the cryostat and seismic adequacy are issues to be resolved.

3. Blanket

3.1 Considerations on materials for blanket

In the previous DEMO design of JAEA (DEMO-2001 [3]), the combination of oxide dispersion strengthened (ODS) steel and supercritical water (25 MPa, 280-510ºC) was chosen to attain high thermal efficiency exceeding 40% [4]. However, when one considers early realization of DEMO, ODS is ruled out from the candidate materials because of its difficulty in fabrication and welding. In contrast, RAFM steel is the most likely option for blanket structural material. Once the blanket material is narrowed down to RAFM, supercritical water is also excluded from the candidate coolant from the point of view of compatibility with RAFM, i.e. concern about corrosion wastage. For this reason, pressurized water is selected as coolant of SlimCS. The surface of the replaceable blanket is coated with tungsten to suppress physical sputtering.

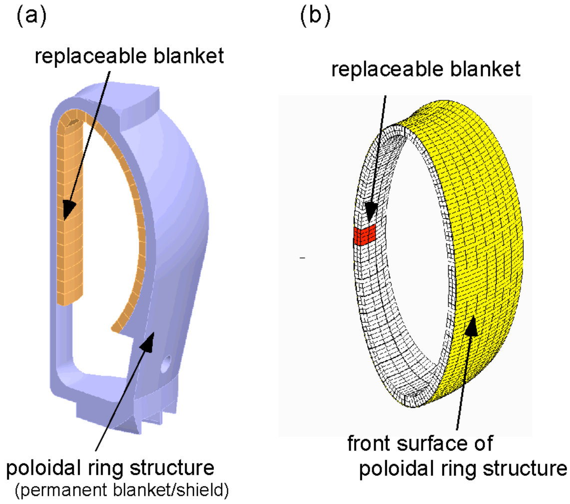


FIG.5. (a) Blanket assembly of SlimCS and (b) mesh data for eddy current and the resulting EM force analysis.

Options for tritium breeder are Li_2TiO_3 and Li_2ZrO_3. Result of neutronics calculations for various blanket models indicates that there is little difference in TBR between Li_2TiO_3 and Li_2ZrO_3. $Be_{12}Ti$ and Be are considered as neutron multiplier. In spite of less TBR, chemical stability of $Be_{12}Ti$ is still attractive in the DEMO design in that it does not react with hot water in case of breaking of coolant boundary. In addition, use of $Be_{12}Ti$ allows a simplified blanket structure because a tight separation of a breeding material such as Li_2TiO_3 from $Be_{12}Ti$ is not required. In this sense, assessment of applicability of $Be_{12}Ti$ to DEMO is one of important tasks in the design study.

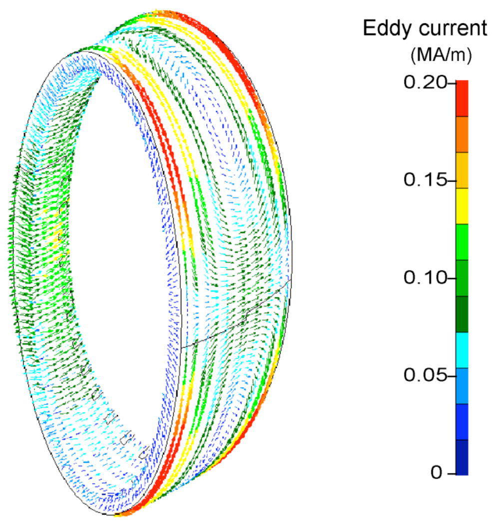


FIG.6. Eddy current when 1.6 m × 0.6 m blanket modules are mounted.

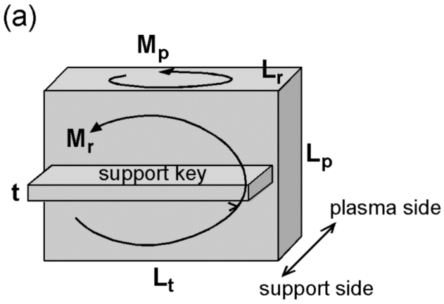
3.2 Segmentation of blanket

A critical issue in the blanket design on DEMO is to assure robustness of the blanket casing and its support against disruptions. This requirement is a difficult matter especially for the replaceable blanket. Because it must not only withstand enormous electromagnetic (EM) forces acting on disruptions, but be easily de-installable and installable for periodic replacement. Generally speaking, while larger blanket casing is desirable in terms of TBR, it is problematic regarding robustness against disruption. After all, the blanket casing should be large as possible on the condition that it withstands disruptions.

In order to determine a reasonable blanket casing in terms of the EM forces, eddy current due to a disruption and the resulting EM force moments were estimated. The replaceable blanket of SlimCS is installed on the poloidal ring structure, which is composed of the permanent blanket/shield on the low field side and the shield on the shield, as shown in FIG.5 (a). For the analysis, the blanket system shown in FIG.5 (a) is modeled with mesh data shown in FIG.5 (b). The replaceable blanket modules are mounted on the cylindrical plate corresponding to the front surface of the poloidal ring structure in the modeling. The dimension of the replaceable blanket is changed in a wide range to study the size dependence of the EM force moments; a toroidal length of 1-2 m and a poloidal length of 0.3-1.2 m. It is assumed that the plasma current is quenched in 0.03 s without suffering vertical displacement event (VDE).

Figure 6 shows the calculated eddy current when the replaceable blanket modules with 1.6 m × 0.6 m and the cylindrical plate. The eddy current induced on the blanket on the high field side is 0.1 MA/m. On the cylindrical plate, saddle-shaped current loops are induced and the maximum current is as high as 0.2 MA/m. In the case of disruptions without accompanying VDE, the radial moment (M_r), which is generated by the coupling of the induced eddy current j_r in the radial direction with B_T, dominates the poloidal (M_p) and toroidal moment (M_t). M_r for various sizes of blanket casing is scalable with the blanket dimensions (L_p, L_t and L_r) as shown in FIG.7 (c). This fact indicates that induced eddy current has a resistive characteristic. When the moment is supported with a key structure with the length of L_t as depicted in FIG.7 (a), the key thickness (t) to withstand the shear force is given by $t > 9M_r/S_mL_t^2 \propto L_p/(L_t+L_r)$ so as. This means that toroidally-long blanket casing has an advantage in the viewpoint of support against disruptions.

(a)



(b)

Model	L_t (m)	L_p (m)	L_r (m)
A	2.0	0.5	0.3
B	1.0	0.5	0.3
C	2.0	0.3	0.3
D	1.0	0.3	0.3
E	2.0	1.2	0.3
F	1.0	1.2	0.3

(c)

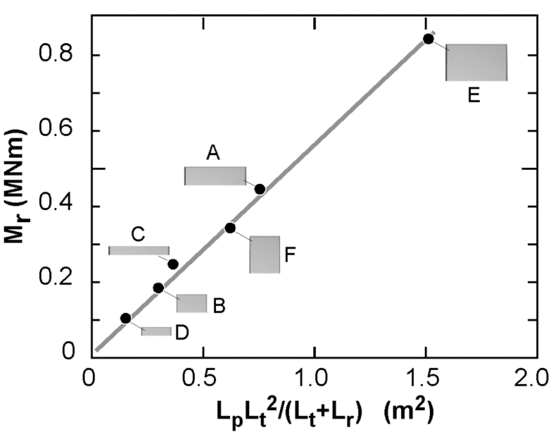


FIG.7. (a) Parameter definition on the replaceable blanket, (b) dimension of each blanket casing for EM force analysis, and (c) scaling of EM force moment M_r.

3.3 Blanket structure

The coverage of blanket is estimated to be 87% of the plasma-facing surface area. The remaining portion is divertor and ports for current drive, diagnostics and fuelling. Considering frames and ribs of the blanket modules and a gap between the neighboring modules, the breeding zone of blanket is reduced to 78% of the plasma-facing surface area. In order to

attain the net TBR of 1.05, which is a requirement taking account of a decrease of tritium due to radioactive decay, a local TBR of 1.35 is necessary. If DEMO is required to supply initial tritium for commercial reactors, the local TBR of 1.38 (equivalent to the gross TBR of 1.08) is necessary to help introducing commercial fusion reactors at the same speed as light water reactors were introduced to market [5]. In the previous DEMO design study (DEMO-2001) with supercritical water solid breeder blanket [4], a local TBR exceeding 1.35 was obtained with a mixture of Li_2TiO_3 and $Be_{12}Ti$ pebbles, which have better compatibility with water at a high temperature than Li_2O and Be. In contrast, the present DEMO (SlimCS) design, pressurized water used as coolant moderates neutrons, reducing TBR. A consideration on heat removal for a neutron wall load of 3 MW/m^2 by pressurized water indicated that the candidate temperature range, pressure and flow velocity of coolant become as follows; i) ΔT = 40K (290-330ºC) at 16 MPa and 7.2 m/s for, ii) ΔT = 60K (300-360ºC) at 22 MPa and 4.8 m/s, and iii) ΔT = 80K (294-374ºC) at 25 MPa and 3.7 m/s, where ΔT is the difference of coolant temperature between inlet and outlet. Since the flow velocity should be 4 m/s or slower in order to ensure a reasonable pressure loss, ΔT should be as high as about 80K. Notice that the PWR conditions of ΔT = 40K are not adequate for such a blanket design.
In the blanket design of DEMO-2001, breeder and multiplier are packed in the form of small pebbles in a layered structure as shown in FIG. 8 (a). The reason why the layers for breeder and multiplier are separated is to avoid reductive degradation of Li_2TiO_3 by Be. An engineering difficulty of the design is how to ensure reliability for the complex partition composed of cooling pipes and plates. In order to resolve the problem, a possible blanket option contains Li_2TiO_3 pebbles in casings made of RAFM as shown in FIG. 8 (b). For neutron multiplication, Be is packed outside the casings for Li_2TiO_3. Another blanket option has a simpler packing structure in which Li_2TiO_3 and $Be_{12}Ti$ pebbles can be packed without separation. A high chemical stability of $Be_{12}Ti$ allows such mixed packing. In this option, the Li_2TiO_3 casings in FIG. 8 (b) are removed from the blanket.

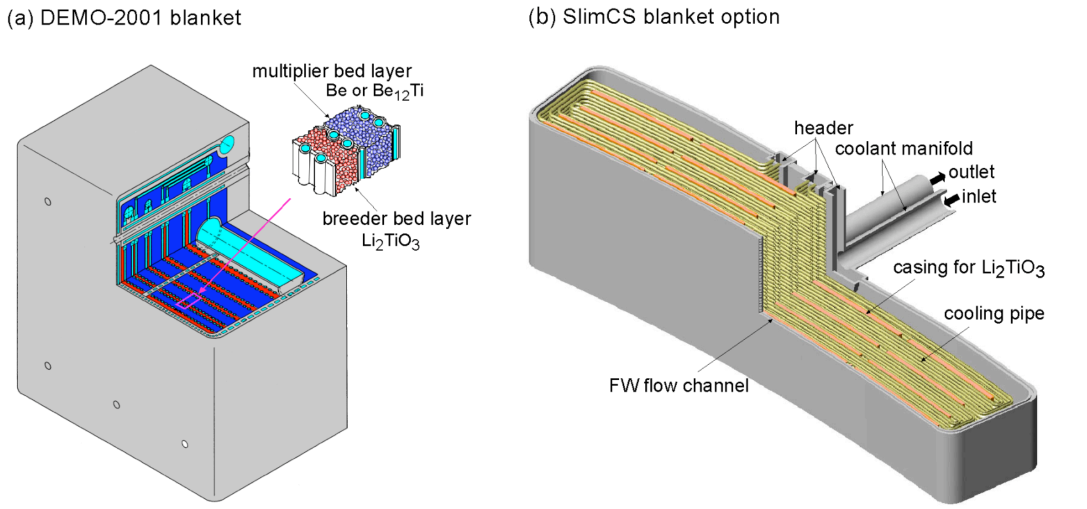


FIG.8. Schematic structure of the replaceable blanket of (a) DEMO-2001 and (b) SlimCS (one of design options).

Detailed layout of the blanket is under study on the basis of neutronics and thermal analysis. A key point of the design is to keep the temperatures of the breeder and multiplier materials in an appropriate range with meeting the required TBR. Result of TBR calculation for several options of blanket materials and structure is summarized in TABLE 2. In the calculation, it is assumed that 90%-enriched ^6Li is used and that neutron wall load is 5 MW/m^2 (peak value of SlimCS). According to the result, one possible option is a combination of Li_2TiO_3 pebbles and Be porous plates, which is anticipated to attain the local TBR of 1.35. A calculated local

TBR of $Li_2TiO_3/Be_{12}Ti$ pebble mixture is 1.31, being slightly lower than the required local TBR. However, this option seems potentially attractive because of ease of fabrication. Although the local TBR is lower than required for considered conditions, the requirement for the local TBR is met when the average neutron wall load (3 MW/m^2) is assumed.

TABLE 2: Local TBR for considered options of blanket materials and structure.

Model		# 1	# 2	# 3	# 4
Breeder	material	Li_2TiO_3 (90%-enriched ^6Li)			
	shape	pebbles			
	temp.limit	900ºC			
Multiplier	material	Be		$Be_{12}Ti$	
	shape	pebbles	plate	pebbles	plate
	temp.limit	600ºC		900ºC	
Packing sturucture		separated		mixed	separated
Local TBR		1.26	1.35	1.31	<1.2
Rating			primary opt.	alternative	

Assumed κ : Be pebbles (7.3 W/mK), Be porous plate (80 W/mK)
Mixed Li_2TiO_3 and $Bi_{12}Ti$ pebbles (4 W/mK)

Another important finding is that a RAFM conducting shell shown in FIG.2 (b) attenuates neutron flux, reducing the TBR. As the calculated TBR of the present blanket options is near the verge of fuel self-sufficiency, a TBR reduction by the conducting shell is relatively important. One must note that a slight increase of TBR by a setback of the shell position can be a trade-off problem between fuel self-sufficiency and the design value β_N.

4. Divertor

4.1 Divertor technology

The divertor plate for SlimCS consists of W monoblock armors and a F82H cooling tube [6]. The original analysis indicated that a surface heat flux of 10 MW/m^2 is removable when the coolant temperature is as low as 200ºC. Then, the maximum surface temperature of W armors is about 1,200ºC, being below the recrystallization temperature (~1,200ºC) and well below the melting point. The maximum temperature of the RAFM cooling tube is 550ºC that is the maximum allowable temperature of F82H to avoid creep damage.

However, the coolant temperature of 200ºC seems unacceptable in the DEMO design from the point of view of material. Because hydrogen peroxide produced by radiation decomposition of water persists in the coolant below 240ºC, which can cause corrosion wastage of the cooling tube. In addition, use of RAFM at such a low temperature is a concern due to irradiation brittleness, albeit lack of test data below 300ºC. To consider these situations, the divertor coolant temperature should be about 300ºC as the coolant inlet temperature of the blanket is. Here, for simplicity, suppose a one-dimensional heat transfer problem via a RAFM plate with a thermal conductivity of 33 W/mK. When the RAFM plate is used in the temperature range of 300-550ºC, the relation of the allowable heat flux (ϕ) and the plate thickness (Δx) is expressed as ϕ [MW/m^2] = $\kappa\Delta T/\Delta x = 8.25/\Delta x$ [mm]. This means that even 1 mm-thick cooling channel can handle with only 8 MW/m^2. As a result, it is reasonable to reduce the allowable divertor heat flux to below 5-8 MW/m^2.

4.2 Divertor simulation

In order to investigate an impact of the heat flux limit set by the above considerations on the physics design, a numerical simulation on divertor was carried out. The outline of the simulation is to seek for divertor plasma conditions to attain the lowest possible heat flux to the divertor plate with the use of D_2 and Ar gas puffing for a given SOL input power (P_{SOL}). It must be noted that, in addition to the heat flux, the divertor plasma should satisfy a divertor electron temperature (T_e^{div}) of less than 10 eV not to suffer serious physical sputtering of the divertor plate. According to Ref. [7], the heat flux of 7 MW/m^2 is obtained for P_{SOL} of 300 MW when D_2 puff of 1×10^{23} s-1, Ar content to fuel ions at the outside divertor plate of 2% and the pumping speed of 200 m^3/s are assumed. Considering a global power balance of the core plasma, P_{SOL} of 300 MW corresponds to P_{fus} of a little lower than 2 GW. When P_{SOL} is increased to 400 MW (corresponding to P_{fus} of about 2.4 GW), the calculated peak heat load becomes as high as 14 MW/m^2. Fortunately, since the simulation does not show any sign of x-point MARFE, further study is likely to find lower divertor heat flux for a given P_{SOL}.

5. Current drive schemes

Candidate options for CD are neutral beam injection (NBI) and waves of electron cyclotron range of frequencies (ECRF). These schemes have both merits and demerits. It should be stressed that each CD scheme has a serious demerit that clouds prospect for DEMO.
In the engineering aspects, ECRF has a lot of advantages; accessibility to the plasma, port size, compatibility with shielding and maintenance, and system efficiency. In the physics aspects, although controllability of plasma current profile is an advantage, low CD efficiency is a major difficulty of ECRF. In the parameter ranges considered for SlimCS, CD efficiency by ECRF (fundamental, O-mode) is between a fourth and a half of that by NBI. Since most of the input power for CD eventually becomes a heat load to the divertor, CD efficiency is a crucial issue in the system design of DEMO.
In addition to an acceptable CD efficiency, NBI has another distinctive advantage in momentum input. When NBI of about 30 MW is at the energy of 0.5 MeV, the plasma rotation of about 3×10^4 m/s is anticipated from a numerical calculation using TASK/TX [8], which exceeds ~$0.003v_A$ being the experimental threshold for RWM stabilization in JT-60U [9] is anticipated. A concern about NBI is susceptibility to Alfvén eigenmodes (AE modes). When AE modes occur in the plasma (perhaps, this is expected in ordinary operational conditions of DEMO), the beam ions suffer anomalous radial transport, resulting in an unexpected current profile and a reduced CD efficiency.

6. Summary

Several design issues were discussed on the basis of recent conceptual design study on DEMO. As long as high beta access, torus configuration and maintenance are concerned, the design outline of DEMO seems to be envisioned with foreseeable technologies. As to TBR of the water-cooled solid breeder blanket, there are a couple of marginal options ensuring tritium self-sufficiency although they do not promise an optimistic outlook. The problem is that neither divertor simulation nor divertor engineering presents effective measures to cope with a divertor heat load originally designed in SlimCS. This will be a common design issue in every DEMO producing a 3 GW-level of fusion output.

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